NON-PUBLIC?: N
ACCESSION #: 8808090251
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Catawba Nuclear Station, Unit 2 PAGE: 1 of 6

DOCKET NUMBER: 05000414

TITLE: Reactor Trip On Steam Generator Low Low Level Due To A Personnel Error
EVENT DATE: 06/26/88 LER #: 88-025-00 REPORT DATE: 07/26/88

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On June 26, 1988, at approximately 2300 hours, an automatic Unit 2 Reactor trip occurred due to low low Steam Generator (S/G) level while the Auxiliary Safeguards Test Cabinet Periodic Test was being conducted by Performance personnel. A Main Steam (SM) Isolation relay was tested to verify closure of the SM Isolation Bypass valves and that the S/G 2D Power Operated Relief Valve (PORV) would change from manual to automatic mode upon an SM Isolation signal. After the S/G SM Isolation Bypass valves were verified to be CLOSED and the S/G PORV control mode was verified, a relay which was activated by this signal was required to be reset per the periodic test. The Performance Test Coordinator (PTC) placed a jumper between two test points to reset the relay. After the relay was reset, the PTC removed the jumper and returned to the Control Room to start another section of the periodic test. The Nuclear Control Operator (NCO) incorrectly assumed that an SM reset was still necessary and depressed the SM1 CLOSED/RESET button to reset SM Isolation. This actuation shut SM1, Main Steam Isolation for S/G D, causing a pressure increase in S/G D and resulting in a level decrease. The Reactor subsequently tripped on S/G D low low level. The Unit was operating at 100% power at the time of the trip.

This incident is attributed to a personnel error. The NCO incorrectly assumed that an SM reset was still necessary and depressed the CLOSE/RESET pushbutton for SM1. This subsequently caused a Reactor trip.

Existing communication methods will be evaluated and improved if necessary. The health and safety of the public were unaffected by this event.

(End of Abstract)

BACKGROUND:

The Reactor (EIIS:RCT) Protection System (EIIS:JC) provides responses to trip the Reactor or actuate appropriate safeguards equipment in time to prevent violation of any of the safety limits. The purpose of PT/2/A/4200/09A, Auxiliary Safeguards Test Cabinet Periodic Test, is to test the operability of the Reactor Protection System final actuating devices without disturbing the operation of the Unit.

The first section of this periodic test, Steam Line Isolation (K623) - Train A, involves an actuation on the Main Steam (EIIS:SB) (SM) System in which the four SM Isolation Bypass valves (EIIS:B) close, and SV19, Steam Generator (EIIS:SG) (S/G) A Power Operated Relief Valve (PORV) control changes from the manual to automatic mode. To perform this section, a signal is generated from the Safeguards Test Cabinet (ESFA) which energizes a Solid State Protection System (EIIS:JC) (SSPS) relay (K623) that in turn actuated an SM relay in cabinet 2SMTC1. The proper operation of the SM valves and PORV control mode are verified. The test signal to the SSPS relay is then reset from ESFA, after which the SM relay (SMAR22) is reset by placing a momentary jumper in 2SMTC1.

DESCRIPTION OF INCIDENT:

On June 26, 1988, at approximately 1900 hours, Performance personnel began the quarterly Auxiliary Safeguards Periodic Test. The Performance Test Coordinator (PTC) briefed the Unit Supervisor on the first section to be performed (K623-Train A), and the Unit Supervisor instructed the PTC to identify Control Room support needed to the Nuclear Control Operator (NCO). At approximately 2230 hours, following a delay due to Secondary System problems, the NCO gave clearance to the PTC to begin the section. The NCO performed the lineup, ensuring that the SM Isolation Bypass Valves were open, and that the S/G PORV control was in manual, at the request of the PTC.

The PTC actuated the SM Isolation signal from the Safeguards Test Cabinet (ESFA), and verified proper SM valve operation. He then reset the test signal to the Solid State Protection System (SSPS) relay (K623) from ESFA, and verified further SM valve operation (that the SM Bypass Valves SM9, SM10, SM11, SM12 remained shut and SV19, S/G A PORV, remained in automatic mode). The PTC then reset the SM relay (SMAR22) in the SM Test

Cabinet (SMTC1, located on 577 elevation, Auxiliary Building) by placing a momentary jumper in the cabinet per the procedure. The PTC then returned to the Control Room to begin the next section of the test.

When the PTC entered the Control Room, the NCO asked him if he had placed the jumper to reset the SM relay. The PTC stated that he had placed the jumper. The NCO noticed that the associated step was also signed off. There was a brief discussion between the NCO and the PTC, but no explicit instruction was given. Based upon his belief that the jumper placed by the PTC was still in place and that this jumper would enable the SM1 CLOSE/RESET pushbutton to operate only as a RESET and not close the valve, the NCO proceeded to depress the CLOSE/RESET pushbutton for SM1.

TEXT:

The NCO depressed the CLOSE/RESET pushbutton for SM1 at 2302 hours. SM1 closed, resulting in a pressure increase in S/G D and a pressure decrease in S/Gs A, B, and C. At 2302:44 hours, SV2, S/G D ASME Coded Safety Relief Valve No. 1, opened. At 2302:45 hours, SV3, S/G D ASME Coded Safety Relief Valve No. 2, opened. At 2302:47 hours, SV4, S/G D ASME Coded Safety Relief Valve No. 3, opened. Also, SV1, S/G D PORV, opened. S/G D level decreased due to increased pressure, and S/G D reached the low low level setpoint and the Reactor tripped automatically on S/G low low level at 2302:47:895 hours. The Main Turbine (EIIS:TBN) tripped on Reactor trip at 2302:47:999 hours. At 2302:48 hours, the Auxiliary Feedwater (EIIS:BA) (CA) Motor Driven pumps (EIIS:P) automatically started on 1 out of 4 S/G Low Low Level (S/G D), and a Blowdown Isolation resulted. A Feedwater Isolation occurred on Reactor trip with low Tave, resulting in decreasing levels in S/Gs A, B, and C. The Condenser (EIIS:COND) Steam Dump valves (Banks 1, 2, and 3) opened as expected.

At 2302:52 hours, the NCO initiated a manual Reactor trip. At 2302:57 hours, the NCO initiated a Turbine trip. At 2302:59:251 hours, S/G A Low Low Level alarm occurred. The CA Turbine Driven Pump started on 2 out of 4 S/G Low Low Level alarms. At 2303:01 hours, S/G B and C Low Low Level alarms occurred. At 2303:19 hours, S/G D PORV (SV1) closed. At 2308 hours, the NCOs reset CA and secured the CAPT to prevent excessive cooldown, and at 2315:13 hours, the NCO reset SM Isolation. At 2344:46:915 hours, the NCOs secured the Main Feedwater (CF) Pump A.

On June 27, 1988, at 0003:01 hours, all S/G low low level alarms were cleared and the Feedwater Isolation valves were realigned. During the Fast Recovery S/G A CF Bypass to CA valve (CA149) did not open when the pushbutton was depressed. Operations initiated Work Request 40756 OPS to repair this valve. Operations realigned S/G Blowdown at 0130 hours, and secured CA Pump

2B at 0203 hours. At 0256 hours, CA149 was repaired and successfully tested.

On June 27, 1988, the Unit entered Mode 2, Startup, at approximately 1228 hours, and entered Unit 1, Power Operation, at approximately 1349 hours.

CONCLUSION:

This incident was attributed to a personnel error. The Nuclear Control Operator (NCO) thought while the jumper was in place, the CLOSED/RESET pushbutton could be depressed without closing SM1. The periodic test did not require or indicate that the NCO was to press CLOSE/RESET on SM1.

Other sections of PT/2/A/4200/09A require Operator action to complete the SM reset process. However, the NCO stated that this was not a factor in this incident.

This incident has been classified as a recurring event since there have been several previous ESF actuations due to inappropriate personnel actions. However, there is no similarity between this Reactor trip and the previous trips.

Shortly after the Reactor trip the Reactor Coolant (EIIS:AB) (NC) Pump 2C seal leakoff flow was lost. This recurring problem will be investigated by Maintenance.

During the Unit Fast Recovery, CA149 did not open when the NCO depressed the pushbutton. An Operations work request was initiated, the valve was repaired and successfully tested. Two previously written Exempt Change Variation Notices (CEVNs) were consequently put on a higher priority. These CEVNs will be scheduled for the next refueling outages and should repair the problems on the CF Bypass to CA valves on both Units.

CORRECTIVE ACTION:

SUBSEQUENT

(1) CA149 was repaired under Operations Work Request 40756 OPS.

(2) Operations stabilized the Unit in Mode 3, Hot Standby, after entering the appropriate emergency procedure.

(3) Operations Management has changed Control Room practices to require greater interface between the Licensed Operators and Testing personnel during all non-emergency activities in order to improve

communication.

(4) Operations has evaluated and initially implemented an enhanced method of communication among Control Room personnel. This will involve conferring with another Operator prior performing non-emergency control board manipulations.

PLANNED

(1) The performance of NC Pump 2C seal leak off flow will be investigated. Flow cannot be established at low pressure.

(2) CEVNs 1809 and 1810 will be scheduled to modify the actuations on CF to CA Bypass valves.

(3) Duke Power personnel will ensure that corrective actions for all inadequate response items identified in the post trip review are developed.

(4) Existing communication methods between Operations and Performance will be evaluated to determine if a more formalized method is necessary.

SAFETY ANALYSIS:

The Reactor trip was automatically initiated from 100% full power due to S/G D low-low level. Feedwater Isolation was automatically initiated upon Reactor trip with low Tave (564 degrees F). Both Motor Driven CA pumps were autostarted upon S/G D low-low level, and a Turbine Driven CA Pump Autostart signal occurred

approximately 11 seconds later upon low-low level in 2 out of 4 S/Gs. The redundant steam supply valves for the Turbine Driven CA pump, SA2 and SA5, opened within 9 seconds of the autostart signal. The Reactor Trip Breakers opened within 74 milliseconds of the SSPS Trip signal, and all of the control rods fell to the bottom of the core, reducing power to decay heat level. Manual Reactor trip was initiated 4 seconds after automatic trip.

Immediately post trip, Reactor Coolant Loop D temperature increased 7 degrees F to a maximum of 598 degrees F. Reactor Coolant temperature in all four loops then decreased and stabilized at 548 degrees F, 9 degrees F from the no-load target of 557 degrees F. Reactor Coolant System pressure increased immediately post trip to a maximum of 2265 psig, and then decreased to a minimum of 1965 psig. Reactor Coolant System pressure then increased and stabilized at 2240 psig within 30 minutes post trip, 5 psi from the no load target of 2235 psig. Pressurizer level decreased to a minimum value of 22% following the trip, and stabilized at 30% within 30 minutes post trip, 5% from the no-load target of 25%. Loop D steam pressure increased to a maximum value of 1200 psig upon closure of the Main Steam Isolation valve (SM1) and Reactor trip. Within 30 minutes post trip, steam pressure in loops A, B, and C had stabilized at 1000 psig, 90 psi from the no-load target of 1090 psig. Loop D steam pressure stabilized at 1015 psig, 75 psi from the no-load target of 1090 psig. S/Gs A, B, C and D reached a minimum wide range indicated value of 44%, 44%, 46%, and 41%, respectively. Steam pressure correction of these values yields actual levels of 57%, 57%, 58%, and 52% for S/Gs A, B, C, and D, respectively.

S/G D Main Steam Isolation Valve (MSIV) SM1 closed in less than the required 5 seconds. As steam pressure increased immediately upon SM1 closure, all three banks of steam dump to condenser valves (with the exception of SB24, which was isolated), the S/G D PORV (SV1), and the S/G D Code Safety Relief valves (SV2, SV3, and SV4) automatically opened to dump steam. Due to SM1 closure, S/G D steam was not dumped to the condenser, but the pressure transient was mitigated by steam relief through the PORV and code safety reliefs. The Auxiliary Feedwater flow rate was well above the 450 gpm minimum cumulative flow to the S/Gs as required by the Reactor Trip Response Emergency Procedure.

Approximately 21 seconds after Reactor trip was initiated, the "approaching loss of adequate subcooling" computer point alarm was generated. The computer program which calculates subcooling utilizes Reactor Coolant System wide range pressure and Reactor Coolant System highest hot leg temperature. A saturation curve, adjusted for measurement uncertainty in the temperature and pressure instrumentation loops, is used to indicate margin of subcooling. The computer point also alarms at 10 degrees F away from actual saturation conditions. Actual saturation conditions were not approached during the post trip response. The minimum Reactor Coolant System subcooling of 19 degrees F occurred at minimum Reactor Coolant System pressure immediately post trip. Subcooling continued to increase and was at 57 degrees F when the computer alarm cleared. Adequate heat sink for core decay heat removal was available and maintained at all times.

Section 15.2.4 of the Catawba FSAR states that inadvertent closure of the MSIV's results in a Turbine trip. The Turbine trip transient is discussed in Section 15.2.3 of the Catawba FSAR. The Reactor trip function via the SSPS (from S/G D low-low level) worked properly, and an overpower delta-T Reactor trip signal was initiated less than one second after the S/G Low-Low Level Trip signal. It should be noted that the Catawba FSAR does not take credit for Auxiliary Feedwater flow since a stabilized plant condition will be reached before Auxiliary Feedwater initiation is normally assumed to occur. In this event, the Auxiliary Feedwater pumps were autostarted at the beginning of the transient due to the S/G D Low-Low Level signal and offsite power being available. It also should be noted that the Catawba FSAR does not assume credit for S/G PORV operation in the turbine trip transient. In addition to the code safety valves, S/G D PORV (SV1) opened to mitigate the pressure increase. Approximately 12.5 minutes post trip, the Main Steam Isolation Bypass valve for D loop, SM9, was opened to provide long-term decay heat removal via steam dump to the condenser. This event is fully bounded by the Turbine trip transient as discussed in Section 15.2.3 of the Catawba FSAR.

All safety related equipment was available throughout this event. The cooldown limits of 100 degrees F per hour for the Reactor Coolant System and 200 degrees F per hour for the pressurizer were not exceeded. Integrity of the fuel cladding, Reactor Coolant System, and Containment structure was maintained at all times.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(iv).

The health and safety of the public was not affected by this incident.

ATTACHMENT # 1 TO ANO # 8808090251 PAGE: 1 of 1

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July 26, 1988

Document Control Desk

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-25

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a)(1) and (d), attached is Licensee

Event Report 414/88-25 concerning a reactor trip on steam generator low low level due to incorrect operator actions during the auxiliary safeguards test cabinet periodic test. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,
/s/ Hal B. Tucker
Hal B. Tucker

JGT/66/sbn
Attachment
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